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1

RELEASE OF FISSION GASES FROM UO₂

by

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B. Lustman

Introduction

The utilization of fuel materials to ever higher burnup levels has introduced a performance limitation which is of relatively minor importance at low burnups, namely the release of insoluble fission products - primarily the noble gases xenon and krypton. In metallic uranium-bearing fuels such gases are not released from the free surface of the fuel-bearing component but appear to be released internally. As a result, a decrease occurs in bulk density, the severity of the decrease increasing with increasing burnup and temperature. However, in the case of uranium oxide no decrease in density has been observed resulting from irradiation, although admittedly no accurate experimental observations have been made on this point. It may be concluded from this observation that the insoluble fission gases can be retained in the oxide structure at least at moderate burnup levels without introducing undue lattice strains. A further observation which confirms this conclusion is that UO₂ may be exposed to high burnup levels at high temperatures without any indication of exaggerated swelling.

It has been observed experimentally that bulk UO₂ fuel elements release fission products when exposed under irradiation to the coolant through defects in experimental sample claddings.¹ Furthermore, clad oxide fuel rods punctured after exposure to pile irradiation release a fraction of the fission product gases generated, the exact value of the fraction depending upon exposure conditions and sample material properties, as will be shown later. All the bulk UO₂ fuel rod specimens thus far examined at Bettis Plant have been fabricated such that the UO₂ is unbonded to the cladding; thus the possibility exists that the observed release of fission gases within the cladding is caused primarily by recoils from surfaces inter-connected with the free volume within fuel rod claddings. However, calculations of the amount of gases liberated by recoils, which take into account the stopping power of gases within fuel rod annuli (helium), probable values for surface energy barriers for emission or absorption of recoil particles, and probable total exposed surface, fail to account quantitatively for the observed rates of fission product gas release by three to six orders of magnitude. Furthermore, the experimentally observed sensitivity of fraction of gas release to operation temperature remains

unexplained if recoils are the principal source of gas release.

It has been concluded, therefore, for the purposes of this paper, that release of fission gases occurs by solid state diffusion and the available data have been so analyzed. By use of these data, the experimentally observed fraction of fission products released has been compared with values calculated from the diffusion rates. Finally the diffusion rate data have been applied to the prediction of pressures developed within uranium oxide fuel elements as a function of burnup, exposure, and heat flux for both a rod and a plate fuel element geometry.

Experimental Data

It must first be recognized that uranium oxide is currently fabricated by cold pressing (extrusion and slip casting have also been applied²) and sintering at temperatures of 1700°C and below. The final structure then consists of a rather fine-grained, single-phase micro-structure containing various proportions of open and closed porosity. The relations between oxide structure, distribution of porosity, density, and fabrication conditions are under investigation by Padden.³ His results indicate that the amount of closed porosity decreases very slowly with increasing density (longer sintering time or higher sintering temperature). The open porosity, however, diminishes to apparently negligible values as measured by relatively gross techniques at densities above 95% of theoretical density (10.96 g/cc). However, measurements performed by Aronson,⁴ utilizing krypton adsorption techniques, indicate that significant amounts of open porosity persist at densities even above 95%. His measurements of variation of surface area with oxide density are plotted in Fig. 1. These data have been utilized to calculate the radius of an equivalent sphere showing the same ratio of surface area-to-volume by the relation: $a = 3 \times D/S$ where a = radius of equivalent sphere in cm, D = fraction of theoretical density, and S = surface area in cm^2/cc . They indicate rather surprisingly that the radius of the equivalent sphere with the same surface area as a pellet of, for example, 95% density is only 4.32×10^{-3} cm. It is apparent, therefore, that the surface area available for release of diffusing substances is far greater than that anticipated from the gross surface area of the specimen. Thus, for all subsequent observations the surface area was estimated from Fig. 1 by using the measured values of sintered density. Note that Fig. 1 was derived from measurements performed upon cold pressed and sintered Mallinckrodt oxide pellets

(hydrogen reduced from UO_3). It is entirely conceivable that this curve is invalid for pellets prepared from different sources of oxide.

Three independent sets of data were used to evaluate the diffusion coefficient of xenon and krypton in UO_2 : data of Booth and Rymer⁵ derived from measurements of Kr-85 release from irradiated UO_2 specimens (prepared from ammonium diuranate precipitated powders) heated at elevated temperatures for a given length of time, data of Susko⁶ also on Kr-85 release from sintered Mallinckrodt oxide pellets utilizing a technique similar to that in reference 5, and finally data of Markowitz and Roll⁷ derived from measurements of rate of Kr-87 and Xe-135 release from oxide specimens exposed in a circulating helium system in the Brookhaven reactor. The diffusion coefficients for the first two sets of data were calculated by utilizing the formula for diffusion from a sphere of constant initial concentration,⁸ the sphere size being chosen for each particular specimen density from Fig. 1. The D values were calculated for the third set of experiments, in which the rate of release of short lived isotopes was measured, by use of a formula which takes into account the rate of isotope decay.⁹ For the third set of data the experimental observations are listed in Table II. In utilizing these data it was assumed that the rates of release at 70 and at 160°C were controlled by processes other than diffusion, and therefore the diffusion contributions at 350 and 550°C were obtained by subtracting the values observed at the two lower temperatures from the high temperature results (in case of Xe-135 only the 550°C value was calculated). The data obtained from the other two experiments are listed in Table I, and the average diffusion coefficients at each temperature are plotted in Fig. 2 as log of D versus reciprocal of absolute temperature.

The indicated variation of diffusion coefficient with temperature shown in Fig. 2 was arbitrarily chosen to be consistent with the values from the third set of data. Even though the results for xenon differ at 550°C by about two orders of magnitude from those for krypton, it was assumed in further analyses that the diffusion coefficient for xenon was the same as that for krypton. It is interesting to note in passing that the relative rates of diffusion noted are consistent with the larger atom size of xenon (1.90 Å) as compared with that of krypton (1.69 Å). Note that both these atoms are much larger in radius than the doubly charged oxygen ion (1.32 Å). No explanation or rationalization can be given for the marked scatter in the data. It is obvious that considerably more

experimentation is required to establish the variation of diffusion coefficient of fission gases with temperature. It is certainly clear, however, that the release of these gases is temperature-sensitive and hence is controlled by mobility in the solid state rather than by surface recoils. The particular variation of diffusion coefficient with temperature shown in Fig. 2 was selected from the many possible methods of fitting the data points since it appears to fit best the subsequently presented data on the release of fission gases from experimental fuel rods during irradiation and has therefore been used in the subsequent analyses.

Also plotted in Fig. 2 are diffusion data derived from measurements of the rate of reaction of UO_2 with oxygen.¹⁰ Note that the activation energy for diffusion of fission gases, 20.2 kcal/mol, is very consistent with the activation energy for oxygen diffusion (21.7 and 26.3 kcal/mole). Auskern¹⁶ has shown that uranium ions diffuse in UO_2 with an activation energy of more than 94 kcal/mole.

It is surprising that the activation energies for the oxygen and krypton diffusional process appears to be very close. It is possible that the mobility of fission gas atoms can be realized only when coupled with oxygen ion mobility and that hence the activation energies actually observed reflect those of oxygen ion diffusivity. The observation that UO_2 can accommodate as much as 12.5 at/o additional oxygen (U_4O_9) without appreciable structural change tends to confirm the assumption that the fission gases can also readily accommodate themselves within the UO_2 structure and that their diffusion occurs by a mechanism related to that of oxygen ions. However, the D_0 value for diffusion of noble gases is lower by six orders of magnitude than that for oxygen. The fact that no change is necessary in the uranium charge to accommodate the neutral noble gas atoms and that the atom sizes of the noble gases are much larger than that of oxygen ions may be factors responsible for this greatly reduced frequency factor.

During the course of evaluation of fuel rods for the FWR, a number of clad specimens of various sintered densities have been exposed in-pile and the amount of fission gases evolved during operation measured radio-chemically by puncturing the fuel rods after their irradiation and collecting the Kr-85 evolved. Because of the relatively large size of these samples and the relatively high heat flux at which they were exposed, a considerable temperature gradient existed from the center to the outside of the fuel pellets. This temperature gradient was calculated from the known values of heat flux utilizing

values of thermal conductivity of UO_2 listed in reference 11. The fuel rod diameter was then divided into five equal volume segments and the temperature considered constant through each segment. The fraction of fission gas release was calculated for each segment from the D values of Fig. 2 by utilizing the following formula derived by Dr. J. M. Markowitz⁷ for diffusion from a sphere with a time-distributed source:

$$F = 1 + \frac{6}{\pi^2 v^2} \sum_{n=1}^{\infty} \frac{1}{n^4} \left(e^{-n^2 \pi^2 v^2} - 1 \right)$$

where F is the fractional release, $v = \sqrt{Dt/a}$, t = time in seconds, a = radius of sphere in cm, D = diffusion coefficient in cm^2/sec . In estimating the temperatures, correction was made for the fact that about a 1-mil annulus of helium gas at 1-atmosphere pressure exists between the fuel and the cladding. Again the sphere of equivalent radius was estimated from Fig. 1 utilizing the measured density of the oxide compact. In Table III are tabulated the available measured fractional releases of Kr-85 together with the calculated values. Note that the agreement for all except specimen 79 S was within a factor of 3, which is considered very reasonable in view of the uncertainties connected with the values of thermal conductivity, heat flux, diffusion coefficient, and the measurement of the experimental amounts of release.

It may thus be concluded that the experimental observations appear to be quite consistent with a diffusional mechanism of fission product release. If this mechanism is correct, the amounts of fission gases released will be very sensitive to the density of the oxide, since this will control the length of diffusion path (from Fig. 1), time of irradiation, and temperature of irradiation.

Application of Data

The data presented in Figs. 1 and 2 were utilized to calculate the pressures that would be developed within fuel rod and fuel plate configurations as a function of burnup and heat flux.

Calculation of Pressures Developed in PWR Core 1 Fuel Rod

The configuration chosen for this calculation was that of the PWR Core 1 fuel rod, namely a Zircaloy-2 clad cylindrical rod with an active fuel length of 9-1/2 in. enclosed in Zircaloy-2 tubing, 0.413-in. OD, 0.027-in. wall thickness with oxide pellet diameter of 0.357 in. The calculations were

performed for two oxide densities, namely 94% (PWR Core 1 reference density) and 96% (showing from Fig. 1 twice the equivalent spherical radius of a 94% dense pellet). It was assumed in this calculation that no free volume was available for containment of the gases and that they accommodated themselves within the tubing by plastic deformation of the tubing. Two cases were calculated, namely, that existing under the condition of operation of 2000-psi external pressure and that occurring during system depressurization and an assumed fuel element temperature of 750°F. The formulas and constants used for the pressure and strain calculations are shown in the Appendix. The amount of fission gas released was calculated for each of the several cases considered by dividing the oxide volume into five equal regions of varying temperatures as was done previously in comparing the experimental and calculated values of fission gas release; where temperatures above the melting point of UO_2 resulted, it was assumed that all gas was released from molten regions. The results of the calculations of the pressures developed are shown in Fig. 3, whereas the cladding strains are plotted in Fig. 4 for the two densities considered and for UO_2 burnups of 3000, 10,000, and 50,000 MWD/T- UO_2 .

It may be noted in Fig. 3 that with increasing surface heat flux, the pressures first decrease. This arises from the fact that less time is required to achieve the burnups noted with increasing surface heat flux and therefore less gas is evolved. As the heat flux increases still further, the rise in fuel element temperature both raises the average temperature of the gas and also increases the diffusion rate of gases in the oxide. Hence the pressures developed increase. As the heat flux rises still further, the weakening of the cladding - because of its increased diameter - becomes sufficiently marked that the internal pressures again decrease. Note that the internal pressures in the depressurized condition are, as would be expected, about 2000 psi below those in the pressurized condition except at high amounts of cladding strain.

The variation of cladding strain with heat flux shown in Fig. 4 is similar to that of internal pressure except that the strains vary to a relatively minor extent between cold depressurized and hot pressurized conditions. Furthermore, after the initial decrease in strain with increasing surface heat flux, the cladding strains continue to increase as more and more gases are evolved and as the gas temperature is increased with increasing surface heat flux.

The internal pressures developed under all conditions are well below the bursting pressures of 8300 to 11,000 psi found by internal pressure bursting tests on Zircaloy tubing at 700°F ¹² (note that the effective bursting pressure of the tubing is the difference between internal and external pressures). However, while fuel element performance does not appear to be pressure-limited, strains in the cladding may well limit the ultimate performance capabilities of the fuel element. If the limit of strain to which a thin-walled cylinder under internal pressure can be subjected is chosen as half the uniform strain at the ultimate tensile strength,¹³ it may then be noted in Fig. 4 that at a 50,000 MWD/T burnup a PWR Core 1 type fuel element would be limited to a surface heat flux of about 300,000 Btu/hr-ft² or a burnup rate less than 1 MWD/T-hr. Of course, such burnup capabilities are far in excess of those required by any present or contemplated PWR Core design.

It may be observed in both Figs. 3 and 4 that, at the same heat flux, appreciably lower pressures and strains are developed the higher the oxide density. This arises solely from the decreased inter-connected porosity and hence increased diffusion path illustrated in Fig. 1. In fact, the density of 96% was chosen for calculation to illustrate the effect of doubling the diffusion path encountered with the PWR Core 1 reference density of 94%. The pressures and plastic strains developed can of course be markedly decreased by providing within the fuel element cladding free space into which the gases evolved can expand. It can be estimated that, in order to keep the pressures within a fuel element to less than 5000 psi, a maximum free space during hot operation of 0.0016 in. per 9-1/2 in. length is required for 3000 MWD/T operation, 0.096 in. at 10,000 MWD/T burnup, and 0.81 in. for 50,000 MWD/T burnup. By provision of such free expansion space, the limitations on performance of cylindrical oxide fuel elements imposed by release of gaseous fission products can be reduced to any desired extent. The results of the calculations from which Figs. 3 and 4 were derived are tabulated in Table IV.

Calculation of Pressures Developed in Plate Fuel Elements

The data on fission product release as affected by density shown in Figs. 1 and 2 have been applied to estimation of pressures and strains developed in flat plate fuel elements consisting of oxide plates 0.04 and 0.08 in. thick contained in 1 in. x 1 in. compartments between 0.020 in. thick Zircaloy cover plates. For these calculations it was assumed that the pressures and strains

developed are limited by expansion of the gases released into the free volume produced by elastic deformation of the cladding. For the 0.040 in. thick $\text{Al}_2\text{O}_3\text{-UO}_2$ plates, it was assumed that the ceramic body contained 9 v/o of enriched uranium oxide and that the diffusion rate of fission gases in Al_2O_3 was similar to that in UO_2 ; the pressures and strains produced were estimated for 100% burnup of the U-235 atoms in 3000, 6000, and 10,000 hr. Thermal conductivities of 2 and 1.5 Btu/hr-ft-°F were assumed for the 0.04 in. $\text{Al}_2\text{O}_3\text{-UO}_2$ and 0.08 in. UO_2 plates, respectively. The results of these calculations are tabulated in Table V and indicate that during hot operation the pressures developed are barely greater than 2000 psi, whereas the clad deflections amount to less than 0.0004 in. at the maximum. During cold depressurization, the maximum internal pressures produced are 45 psi and the maximum clad deflection about 0.008 in. Thus this particular plate geometry, which was selected to simulate a possible type of FWR seed configuration, is considered feasible from the point of view of internal pressures developed by fission gas release. The calculations were performed for an inter-connected porosity equivalent to a density of 97% of theoretical. Preliminary experiments by Mr. B. E. Schaner¹⁴ have indicated that such densities are readily obtained by cold pressing and sintering procedures in $\text{Al}_2\text{O}_3\text{-23 w/o UO}_2$ admixtures.

Similar calculations were performed for 1 in. x 1 in. x 0.08 in. UO_2 plates of 95% density chosen to simulate geometries of experimental FWR blanket fuel elements. For these calculations it was assumed that a burnup of 50,000 MWD/T would be obtained in 10,000 hr of operation. The results of these calculations are summarized in Table V. Again, under conditions of pressurized operation, the pressures developed are about 2010 psi and the maximum clad deflection, 0.0019 in. During cold depressurization, on the other hand, internal pressures of 120 psi are developed at clad deflections of 0.016 in. Although these deflections appear excessive, they can be reduced by reduction in plate size (clad deflections decrease almost in proportion to the area of the compartment); by increase in UO_2 density, with a consequent decrease in fraction of gases released; or by decrease of fuel thickness, which not only decreases the amount of fission gas generated but also decreases the temperatures and hence the diffusion rate of the gases out of the oxide structure. As shown in Table V, maximum clad deflections of 0.007 in. are attained by decrease in compartment size to 0.833 in.² and increase in fuel density to 97%.

In summary, therefore: whereas considerable further work must be performed to confirm the estimates of quantities of fission product gases released, there is little question concerning the ultimate feasibility of flat plate configurations for oxide fuels, assuming, of course, that the mechanism of gas release is that proposed in this report.

Proposed Future Work

It appears clear that considerable accurate experimentation must be performed to establish without question the mechanism of fission product release. Many further experiments of the type listed in Table II, where the rate of release of fission product gases is measured from irradiation specimens heated for various lengths of time at various temperatures, are required to establish the variation of diffusion coefficient with temperature for various oxide densities. Similar work must be performed on mixtures of uranium oxide with Al_2O_3 , ZrO_2 , etc. to establish the mobility of gases in such structures.

Additional experiments must be performed in-pile so that the quantities of gas released during pile operation can be compared directly with those released from irradiated specimens on heating after irradiation. Such work is required not only to determine possible effects of the pile environment on diffusion rate of gases but also to establish the existence of mechanisms of gas release other than those considered in this report. Again the relations between oxide density and surface area, oxide structure, and admixture with other materials must be established. Such experiments are presently being performed by Dr. J. M. Markowitz at Brookhaven.

Finally the relations predicted from such measurements must be confirmed by measurements of fission gas release from experimental fuel elements similar to those tabulated in Table III. Such measurements are planned in the course of a statistical irradiation program which is being inserted in one of the Hanford reactors.¹⁵

Summary and Conclusions

The available data on release of fission gases from irradiated uranium oxide were correlated on the basis of a diffusional mechanism of gas mobility in the oxide. The diffusion rates thus estimated were found to vary with temperature and had an activation energy similar to that observed in case of the diffusion of oxygen in UO_2 . The frequency factor, however, was found to

be six orders of magnitude less than that observed for oxygen diffusion. The amounts of fission gas released in experimental fuel rods were found to agree within a factor of 3 with those predicted by such a diffusional mechanism when correction was made for the open porosity or density of the oxide bodies. The pressures and strains developed within fuel rods of the PWR Core 1 configuration and in experimental flat plate fuel elements were calculated for various burnups and burnup rates.

The following conclusions may be drawn from this investigation:

1. The release of fission gases from irradiated UO_2 is in all probability controlled by diffusion of the gases from the oxide matrix.
2. The oxide density is important in establishing the path length for diffusion to external surfaces.
3. The PWR Core 1 blanket fuel element geometry is considered safe with respect to the internal pressures and cladding strains developed to burnups and burnup rates far in excess of those proposed in PWR Core 1.
4. Compartmented flat plate configurations are considered feasible for both seed and blanket fuel elements from the points of view of internal pressures and cladding strains.

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Appendix: Calculation of Fission Product Pressures

1. Pressure Developed in PWR Core 1 Fuel Rod during Operation

$$V = \frac{\pi L}{4} (D^2 - D_o^2)$$

$$D_o = \frac{D}{(1 + \epsilon)}$$

$$V = \frac{\pi L D_o^2 \epsilon}{2}$$

$$VP_i = N*RT$$

$$2\sigma l = (P_i - P_o) D_o (1 + \epsilon)$$

$$\sigma = K\delta^n \cong K\epsilon^n$$

$$\epsilon = \frac{2N*RT}{\pi D_o^2 L P_i}$$

$$\frac{\epsilon^n}{1 + \epsilon} = \frac{(P_i - P_o) D_o}{2Kl}$$

2. Pressure Developed during Cold Depressurization

As above, except $P_o = 0$

3. Pressure Developed in Compartmented Plate during Operation

Assume plate deforms in shape of square based pyramid as a result of internal pressure. Compartments in shape of rigidly fastened squares.

$$V = \frac{2}{3} ya^2$$

$$P_i = \frac{N*RT}{2/3 ya^2}$$

$$y = \frac{0.0138 (P_i - P_o) a^4}{El^3}$$

4. Pressure Developed during Cold Depressurization

As above, except $P_o = 0$ N^* = No. mols fission gas escaping from total volume V = Volume available to contain fission gas L = Length of fuel (9.5 in.) D = Instantaneous diameter of fuel rod D_o = Original fuel rod diameter = 0.387 in. ϵ_o = Cladding strain σ = Cladding stress = $K\delta^n \cong K\epsilon^n = 49,000 \epsilon^{0.038}$ R = Gas constant = $73.8 \frac{\text{lb in.}}{\text{g mol } ^\circ\text{K}}$ T = Gas temperature, $^\circ\text{K}$ = 473 $^\circ\text{K}$ for plates cold= Average UO_2 temperature for rods hot= Maximum UO_2 temperature for plates hot= 673 $^\circ\text{K}$ for rods cold ℓ = Cladding thickness

= 0.027 in. for rod

= 0.020 in. for plate

 P_i = Internal pressure, psi P_o = External pressure = 2000 psi a = Edge length of plate compartment square E = Young's Modulus = 10^7 psi hot = 1.3×10^7 psi cold

TABLE I

ANALYSIS OF FISSION PRODUCT RELEASE DATA (Kr-85) FROM IRRADIATED UO_2

Density % Theo	Radius of Equiv Sphere cm	Temp °C	Heating		D cm ² /sec	Ref
			Time hr	Fractional Release		
75.2	9.45×10^{-5}	1400	4	0.24	3.59×10^{-15}	5
85.3	3.97×10^{-4}	1400	4	0.29	9.65×10^{-14}	5
89.9	9.35×10^{-4}	1400	4	0.22	2.89×10^{-13}	5
96.4	7.29×10^{-3}	1400	4	0.06	1.062×10^{-12}	5
96.7	8.2×10^{-3}	1400	4	0.02	1.312×10^{-13}	5
97.2	1.039×10^{-2}	1400	4	0.02	2.11×10^{-13}	5
85.3	3.97×10^{-4}	1600	4	0.50	3.39×10^{-13}	5
89.9	9.35×10^{-4}	1600	4	0.36	8.75×10^{-13}	5
95.4	4.54×10^{-3}	1600	4	0.16	3.43×10^{-13}	5
80	1.74×10^{-4}	250	2	0.00026	2.28×10^{-20}	6
		600	5	0.00027	0.985×10^{-20}	6
		1000	5	0.00094	1.183×10^{-19}	6
91	1.28×10^{-3}	250	1.5	0.000071	1.22×10^{-19}	6
		600	6	0.00073	3.23×10^{-18}	6
		1000	6	0.00076	3.51×10^{-18}	6
91	1.28×10^{-3}	370	2.5	0.0003	1.3×10^{-18}	6
		600	8	0.0003	9.07×10^{-19}	6
		1000	5.5	0.0026	4.46×10^{-17}	6
95	4.32×10^{-3}	250	100	0.0000033	4.72×10^{-22}	6
		600	24	0.000091	1.391×10^{-19}	6
		1000	24	0.00051	4.37×10^{-18}	6
91.8	1.52×10^{-3}	1000	48	0.0003	0.94×10^{-19}	6
92.6	1.88×10^{-3}	1000	48	0.0005	3.97×10^{-19}	6
92.2	1.67×10^{-3}	1000	48	0.0004	2.02×10^{-19}	6
91.7	1.51×10^{-3}	1000	48	0.0002	4.15×10^{-20}	6

TABLE II
 DIFFUSION OF XENON AND KRYPTON FROM UO_2 IN-PILE AS MEASURED IN
 BROOKHAVEN FISSION GAS RELEASE EXPERIMENT⁷

Temp °C	Density % Theo	Radius of Equiv Sphere cm	Isotope	Fraction Escaping sec ⁻¹	Diffusion Coefficient cm ² /sec
350	97.5	1.37×10^{-2}	Kr ⁸⁷	0.275×10^{-4}	2.1×10^{-18}
550	97.5	1.37×10^{-2}	Kr ⁸⁷	2.07×10^{-4}	1.19×10^{-16}
550	97.5	1.37×10^{-2}	Xe ¹³⁵	5.61×10^{-5}	1.14×10^{-18}

20

TABLE III

COMPARISON OF OBSERVED AND CALCULATED FRACTIONAL RELEASE
OF Kr-85 FROM IRRADIATED FUEL RODS

Specimen	Fuel Diam in.	Irradiation Time sec	Burnup MWD/T	Density % Theo	Radius Equiv Sphere cm	Center Temp °C	Heat Flux Btu/hr-ft ²	Fractional Release	
								Observed	Calculated
X-1-G-3	0.357	4.85×10^6	2780	95	4.32×10^{-3}	1815	353,000	23.10^{-2}	11.8×10^{-2}
X-1-G-5	0.357	4.85×10^6	2060	95	4.32×10^{-3}	1250	261,000	16.10^{-2}	4.71×10^{-2}
X-1-B-4	0.242	3.63×10^6	1650	80	1.742×10^{-4}	760	147,000	9.5×10^{-2}	24.6×10^{-2}
X-1-C-6H	0.3444	5.61×10^5	350	95	4.32×10^{-3}	925	241,000	5.8×10^{-3}	7.11×10^{-3}
WAPD-25-L-2	0.357	1.58×10^6	815	95	4.32×10^{-3}	1600	315,000	1.6×10^{-2}	2.8×10^{-2}
X-1-D-79 S	0.3444	8.67×10^6	5870	95	4.32×10^{-3}	925	241,000	1.1×10^{-3}	27.9×10^{-3}

21

TABLE IV

CALCULATION OF PRESSURES AND STRAINS DEVELOPED IN PWR CORE 1 FUEL RODS*

<u>Irrad time hr</u>	<u>Heat Flux Btu/hr-ft²</u>	<u>Oxide Density %</u>	<u>Mols Fission Gas Released</u>	<u>Gas Temp** °K</u>	<u>P_{ext} psi</u>	<u>P_{int} psi</u>	<u>Cladding Strain</u>	<u>Free length to keep P_{int} below 5000 psi in.</u>
Burnup: 3,000 MWD/T-UO ₂								
3,000	169,600	94	0.0000165	875	2000 0	6750 4790	7.04 x 10 ⁻⁵ 7.13 x 10 ⁻⁵	0.00164
		96	0.0000082	875	2000 0	6620 4650	3.55 x 10 ⁻⁵ 3.9 x 10 ⁻⁵	
5,000	101,800	94	0.0000093	761	2000 0	6620 4680	3.5 x 10 ⁻⁵ 4.37 x 10 ⁻⁵	0.00092
		96	0.0000046	761	2000 0	6500 4550	1.78 x 10 ⁻⁵ 2.24 x 10 ⁻⁵	
10,000	50,900	94	0.0000053	673	2000 0	6510 4580	1.8 x 10 ⁻⁵ 2.56 x 10 ⁻⁵	0.00053
		96	0.0000026		2000 0	6390 4470	0.916 x 10 ⁻⁵ 1.31 x 10 ⁻⁵	
∞	0	94	0.00049	673	2000 0	7330 5420	0.00148 0.002	0.00486
		96	0.00049		2000 0	7330 5420	0.00148 0.002	

* Fuel Rods 0.413 in. OD x 0.027 in. wall x 0.357 in. UO₂ diameter x 9.5 in. long

**Average Gas Temperature assumed to be 673°K for cold, depressurized condition

TABLE IV (Cont)

CALCULATION OF PRESSURES AND STRAINS DEVELOPED IN PWR CORE 1 FUEL RODS*

<u>Irrad time hr</u>	<u>Heat Flux Btu/hr-ft²</u>	<u>Oxide Density %</u>	<u>Mols Fission Gas Released</u>	<u>Gas Temp** °K</u>	<u>P_{ext} psi</u>	<u>P_{int} psi</u>	<u>Cladding Strain</u>	<u>Free length to keep P_{int} below 5000 psi in.</u>
Burnup: 10,000 MWD/T-UO ₂								
3,000	565,000	94	0.00097	2,253	2000	7760	0.00925	0.0961
					0	5560	0.00385	
		96	0.00078		2000	7740	0.00746	
					0	5510	0.00313	
5,000	339,000	94	0.000445	1,350	2000	7480	0.00264	0.0441
					0	5410	0.00181	
		96	0.000246		2000	7330	0.00149	
					0	5290	0.00103	
10,000	169,600	94	0.000092	875	2000	7050	0.000376	0.0091
					0	5110	0.000397	
		96	0.000046		2000	6930	0.000192	
					0	4980	0.000204	
∞	0	94	0.00163	673	2000	7610	0.00473	0.1618
					0	5670	0.00635	
		96	0.00163		2000	7610	0.00473	
					0	5670	0.00635	

* Fuel Rods 0.413 in. OD x 0.027 in. wall x 0.357 in. UO₂ diameter x 9.5 in. long

**Average Gas Temperature assumed to be 673°K for cold, depressurized condition

23

TABLE IV (Cont)

CALCULATION OF PRESSURES AND STRAINS DEVELOPED IN PWR CORE 1 FUEL RODS*

<u>Irrad time hr</u>	<u>Heat Flux Btu/hr-ft²</u>	<u>Oxide Density %</u>	<u>Mols Fission Gas Released</u>	<u>Gas Temp** °K</u>	<u>P_{ext} psi</u>	<u>P_{int} psi</u>	<u>Cladding Strain</u>	<u>Free length to keep P_{int} below 5000 psi in.</u>
Burnup: 50,000 MWD/T-UO ₂								
3,000	2,830,000	94	0.00815	21,473	2000	5900	0.975	0.808
					0	5930	0.0304	
		96	0.00815		2000	5900	0.975	
					0	5930	0.0304	
5,000	1,765,000	94	0.0077	10,583	2000	7000	0.384	0.764
					0	5920	0.0287	
		96	0.00757		2000	7020	0.376	
					0	5920	0.282	
10,000	848,000	94	0.0068	3,394	2000	7740	0.098	0.674
					0	5910	0.0254	
		96	0.00626		2000	7750	0.09	
					0	5900	0.0234	
20,000	424,000	94	0.00428	1,632	2000	7850	0.0293	0.31
					0	5870	0.0162	
		96	0.00319		2000	7840	0.0218	
					0	5820	0.0122	

* Fuel Rods 0.413 in. OD x 0.027 in. wall x 0.357 in. UO₂ diameter x 9.5 in. long

**Average Gas Temperature assumed to be 673°K for cold, depressurized condition

WAPD-173

24

TABLE IV (Cont)

CALCULATION OF PRESSURES AND STRAINS DEVELOPED IN PWR CORE 1 FUEL RODS*

<u>Irrad time hr</u>	<u>Heat Flux Btu/hr-ft²</u>	<u>Oxide Density %</u>	<u>Mols Fission Gas Released</u>	<u>Gas Temp** °K</u>	<u>P_{ext} psi</u>	<u>P_{int} psi</u>	<u>Cladding Strain</u>	<u>Free length to keep P_{int} below 5000 psi in.</u>
Burnup: 50,000 MWD/T-UO ₂ (Cont)								
100,000	84,800	94	0.000398	723	2000	7290	0.0013	0.0395
					0	5390	0.00164	
		96	0.000199		2000	7160	0.000661	
					0	5260	0.00084	
∞	0	94	0.00815	673	2000	7840	0.023	0.808
					0	5930	0.0304	
		96	0.00815		2000	7840	0.023	
					0	5930	0.0304	

* Fuel Rods 0.413 in. OD x 0.027 in. wall x 0.357 in. UO₂ diameter x 9.5 in. long

**Average Gas Temperature assumed to be 673°K for cold, depressurized condition

TABLE V

PRESSURES AND STRAINS DEVELOPED IN COMPARTMENTED FLAT PLATE FUEL ELEMENTS

Comp	Fuel		Compartment Size in.	Radius Equiv Sphere cm	Time, hr		Frac Release F.G.	Surface Heat Flux Btu/hr-ft ²	P _{ext} psi	P _{int} psi	Clad Deflec in.
	Thick- ness in.	Density % Theo			100% Burnup	Remaining at 400°C					
Al ₂ O ₃ + 21 w/o UO ₂	0.040	97	1 x 1	0.01031	3,000	7000	0.01241	1,020,000	2000 0	2002.1 45	0.000363 0.0078
	0.040	97	1 x 1	0.01031	6,000	4000	0.00644	510,000	2000 0	2000.9 33	0.000158 0.0058
	0.040	97	1 x 1	0.01031	10,000	0	0.00344	306,000	2000 0	2000.4 24	0.000075 0.004
UO ₂	0.080	95	1 x 1	0.00432	10,000	-	0.0355	442,000	2000 0	2010.8 120	0.00186 0.016
UO ₂	0.080	95	0.833 x 0.833	0.00432	10,000	-	0.0355	442,000	2000 0	- 173	- 0.011
UO ₂	0.080	97	1 x 1	0.01031	10,000	-	0.01486	442,000	2000 0	2004.5 78	0.00078 0.0103
UO ₂	0.080	97	0.833 x 0.833	0.01031	10,000	-	0.01486	442,000	2000 0	- 112	- 0.0071

26

Fig. 1 Variation of Surface Area with UO₂ Density
(Neg 20429-7)

Fig. 2 Diffusion Rates of Xe and Kr in UO_2
(Neg 20429-3)

(Neg 20429-6)

Fig. 4 Plastic Strain Developed in PWR Cladding (No Expansion Space)

(Neg 20429-5)